November 26, 2002

Mr. Michael Kansler
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: 1.4 PERCENT POWER UPRATE (TAC NO. MB5297)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 213 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated May 30, 2002, as supplemented on September 13 and November 6 and 20, 2002.

The amendment revises the facility operating license and the TSs to increase the licensed core thermal power level to 3067.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3025 MWt. The power uprate is based on the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, "ECCS Evaluation Models," to Part 50 of Title 10 of the *Code of Federal Regulations*. Specifically, the reduced uncertainty is obtained by using a more accurate measurement of feedwater flow.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures:
1. Amendment No. 213 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

November 26, 2002

Mr. Michael R. Kansler
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: 1.4 PERCENT POWER UPRATE (TAC NO. MB5297)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 213 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated May 30, 2002, as supplemented on September 13 and November 6 and 20, 2002.

The amendments revises the facility operating license and the TSs to increase the licensed core thermal power level to 3067.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3025 MWt. The power uprate is based on the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, "ECCS Evaluation Models," to Part 50 of Title 10 of the *Code of Federal Regulations*. Specifically, the reduced uncertainty is obtained by using a more accurate measurement of feedwater flow.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 213 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

File Name: C:\ORPCheckout\FileNET\ML023290636.wpd

ACCESSION NO: ML023290636 *See previous concurrences

OFFICE	PDI-1/PM	PDI-1/LA	SPSB/SC*	SRXB/SC*	SPLB/C*	EMCB/C*	EMEB/SC*
NAME	PMilano	SLittle	JLee for MReinhart	FAkstulewicz	JHannon	WBateman	KManoly
DATE	11/25/02	11/25/02	11/19/02	11/20/02	11/21/02	11/21/02	11/20/02
OFFICE	IEHB/C*	EEIB/C*	RLEB/SC*	OGC*	PDI-1/SC	PDI/D	DLPM/D
NAME	TQuay	JCalvo	JTappert	AHodgdon	RLaufer	SRichards	JZwolinski
DATE	11/20/02	11/21/02	11/19/02	11/22 /02	11/25/02	11/25/02	11/25/02

OFFICIAL RECORD COPY

DATED: <u>November 26, 2002</u>

AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT UNIT 3

PUBLIC
PDI R/F
R. Laufer
S. Little
P. Milano
OGC
G. Hill (2)
W. Beckner
C-Y Liang
L. Lois
C. Khan
T. McLellan
B. Marcus
N. Trehan
C-I Wu
M. Hart
ACRS
B. Platchek, R-I

cc: Plant Service list

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 213
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated May 30, 2002, as supplemented on September 13 and November 6 and 20, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by SRichards for/

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 26, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 213

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1.1-5	1.1-5
2.0-2	2.0-2
3.3.2-8	3.3.2-8
3.3.2-9	3.3.2-9
3.3.2-11	3.3.2-11
3.4.3-3	3.4.3-3
3.4.3-4	3.4.3-4
3.4.3-5	3.4.3-5
3.4.12-9	3.4.12-9
3.4.12-10	3.4.12-10
3.4.12-11	3.4.12-11
3.4.12-12	3.4.12-12
3.7.1-3	3.7.1-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated May 30, 2002, as supplemented by letters dated September 13 and November 6 and 20, 2002, Entergy Nuclear Operations, Inc. (ENO or the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs). The requested changes would revise the facility operating license and the TSs to reflect a 1.4 percent (%) increase in the reactor core thermal power level from 3025 megawatts thermal (MWt) to 3067.4 MWt. The September 13, November 6, and November 20 letters provided clarifying information that did not enlarge the scope of the amendment request or change the initial proposed no significant hazards consideration determination.

1.1 Background

Reactor core thermal power is validated by a nuclear steam supply system (NSSS) energy balance (calorimetric) calculation. The reliability of this calculation depends primarily on the accuracy of feedwater flow, temperature, and pressure measurements. Because the measuring instruments have measurement uncertainties, margins are included to ensure the reactor core thermal power levels do not exceed safe operating levels.

When the initial IP3 operating license was issued in 1976, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, “ECCS [emergency core cooling system] Evaluation Models,” required licensees to assume a 2.0% measurement uncertainty for the reactor thermal power and to base their transient and accident analyses on an assumed power level of at least 102% of the licensed thermal power level. The 2% power margin was intended to address uncertainties related to heat sources and measuring instruments’ accuracy. Appendix K to 10 CFR Part 50 did not allow any credit for demonstrating that the measuring instruments may be more accurate than originally assumed in the ECCS rulemaking. Thus, Appendix K did not originally require the power measurement uncertainty be determined, but instead required a fixed 2% margin.

On June 1, 2000, the Nuclear Regulatory Commission (NRC) published a final rule (65 FR 34913) that allows licensees to justify a smaller margin for power measurement uncertainty when more accurate instrumentation is used to calculate the reactor thermal power and calibrate the neutron flux instrumentation. This revision to Appendix K to 10 CFR Part 50 allows licensees to use a power uncertainty of less than 2% in the design-basis loss-of-coolant accident (LOCA) analyses, provided that state-of-the-art feedwater flow measurement devices that provide for a more accurate calculation of power are utilized. License amendments to increase power based on improved feedwater flow measurements are commonly referred to as measurement uncertainty recapture power uprates.

1.2 <u>Proposed IP3 Measurement Uncertainty Amendment</u>

In support of its May 30 application, the licensee performed a re-evaluation of various NSSS parameters, safety-related systems and components, nuclear fuel, and accident analyses related to operation at the increased reactor power level of 3067.4 MWt. The licensee's application is based on a reduced core-thermal-power uncertainty because of a more accurate measurement of feedwater flow. The improved accuracy is achieved by installation of the Caldon Leading Edge Flow Meter (LEFM) Check System. The improved flow measurement instrumentation would allow the licensee to operate IP3 with a margin below the 2.0% margin previously used in the licensing basis ECCS analyses. The licensee stated that, as a result of the improvement in the flow measurement accuracy, the IP3 power measurement uncertainty has been reduced from 2.0% to 0.6%.

ENO's submittals referenced Caldon Engineering Reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Using the LEFM Check System," and ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM Check System," to provide a generic basis for the proposed 1.4% power uprate. Engineering Reports ER-80P and ER-160P were approved by the staff, respectively in safety evaluation reports (SERs) dated March 8, 1999, and January 19, 2001.

2.0 <u>REGULATORY EVALUATION</u>

The NRC staff finds that ENO in its May 30, 2002, application addressed the applicable regulatory requirements. 10 CFR establishes the fundamental regulatory requirements with respect to the safety-related systems. The regulatory standards on which the staff bases its acceptance are as follows:

a. 10 CFR 50.46, "Acceptance Criteria for [ECCSs] for light-water nuclear power reactors," requires, in part, that the ECCS cooling performance be calculated in accordance with an acceptable evaluation model and for a number of postulated LOCAs. Comparisons to experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed.

b. Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 sets forth the requirements for the models.

c. Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 provides, in part, requirements related to the establishment of reactor pressure vessel (RPV) pressure versus temperature (P-T) limit curves for any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. Appendix G also references, via 10 CFR 50.55a, "Codes and standards," the requirements prescribed in American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G.

d. Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50 sets forth requirements related to the establishment of a facility’s RPV surveillance capsule program and withdrawal schedule. Appendix H also references the guidance in American Society for Materials and Testing Standard Practice E185, “Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.”

e. ASME Code Sections III and XI provide additional guidance regarding the evaluation of the structural integrity of RPV internals and ASME Code Class 1, 2, and 3 components, component supports, and core support structures.

f. 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” requires that a review of the safety-related equipment should be performed to assure the adequacy of the equipment qualification for the normal and accident conditions expected in the area where the equipment is located.

g. 10 CFR 50.61, “Fracture toughness requirements for protection against pressurized thermal shock events,” provides criteria for assessing events or transient that could cause severe overcooling of the reactor coolant system with or followed by significant pressure in the RPV.

h. 10 CFR 50.63, “Station Blackout,” requires that all nuclear power plants must have the capability to withstand a loss of all AC power for an established period of time, and to recover therefrom.

i. 10 CFR Part 100, “Reactor Site Criteria,” establishes, in part, criteria important to assuring that radiological dose from normal operation and postulated accidents will be acceptably low.

j. Several General Design Criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

    1. GDC-4, “Environmental and dynamics effects design basis,” requires that systems, structures, and components, be able to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These systems, structures, and components shall also be appropriately protected for dynamic effects such as missiles, pipe whipping and discharging fluid that result from system failures and from events and conditions outside the plant.

    2. GDC-14, “Reactor coolant pressure boundary,” requires the boundary to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

    3. GDC-17, “Electric Power Systems,” requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The safety function

for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that the containment integrity and other vital functions are maintained in the event of postulated accidents.

4. GDC-19, “Control Room,” requires, in part, that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions.

5. GDC-30, “Quality of Reactor Coolant Pressure Boundary,” requires, in part, that means be provided for detecting and, to the extent practical, identifying the location of reactor coolant leakage.

6. GDC-31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” requires that the boundary be designed with sufficient margin.

k. <u>NRC Regulatory Guides (RGs)</u>

1. RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” March 2001.

2. RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” provides guidance for the NRC staff’s review of RPV P-T limit curves.

3. RG 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy [USE] Less Than 50 Ft-Lb,” along with Appendix K to Section XI of the ASME Code provide guidance when USE equivalent margins analyses are required.

l. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.”

m. NUREG-1061, Volume 3, “Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks,” and draft Standard Review Plan Section 3.6.3, “Leak Before Break [LBB] Evaluation Procedures,” provide guidance for evaluating the technical basis for a licensee’s application of LBB.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>Feedwater Flow Measurement Technique and Power Measurement Uncertainty</u>

<u>Power Calorimetric Instrumentation</u>

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system. This calculation is called the "secondary calorimetric" for a pressurized-water reactor. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow and feedwater net enthalpy measurements. Thus, an accurate measurement of feedwater flow and temperature will result in an accurate calorimetric calculation and an accurate calibration of the nuclear instrumentation.

The instrumentation for measuring feedwater flow typically uses a venturi meter, an orifice plate, or a flow nozzle to generate a differential pressure proportional to the feedwater velocity in the pipe. Of these three differential pressure devices, a venturi meter is most widely used for feedwater flow measurement in nuclear power plants. The feedwater temperature is typically measured by resistance temperature detectors (RTDs). The IP3 design uses a flow nozzle for flow and thermocouples for temperature measurement in each of the four feedwater lines. Similar to the venturi, the major advantage of a flow nozzle is the relatively low head loss created as the fluid passes through the device. The major disadvantage of the flow nozzle is fouling, which causes the meter to indicate a higher differential pressure and hence a higher than actual flow rate. Since feedwater flow rate is directly proportional to calorimetric power, this error in feedwater flow rate measurement leads the plant operator to calibrate the nuclear instrumentation at a higher than actual core power. Calibrating the nuclear instrumentation to indicate higher than actual core power is conservative with respect to reactor safety, but causes the generation of electrical power proportionately lower when the plant is operated at its indicated thermal power rating. To eliminate the effects of nozzle fouling, the flow nozzle device must be removed, cleaned, and re-calibrated. The high cost of flow meter calibration and the desire to improve flow instrumentation accuracy prompted the nuclear industry to assess other flow measurement techniques. The use of an ultrasonic flow meter implementing transit time technology was found to be a viable alternative.

The Caldon LEFM is an ultrasonic flow meter, using acoustic energy pulses to determine the feedwater mass flow rate and temperature. This system is based on time-of-flight (transit time or counter-propagation) technology. The transit time technology sends an ultrasonic signal diagonally through the fluid and then measures the time it takes to travel upstream and downstream. The sound travels faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The difference in these times is proportional to the velocity of the fluid in the pipe. The LEFM uses these transit times to determine the fluid velocity and temperature (the temperature of the fluid is determined from a predetermined correlation between the fluid pressure and sound velocity in the fluid).

At IP3, the LEFM Check system consists of an electronics cabinet, with four channels of instrumentation and four measurement sections (spool pieces) permanently installed, one spool piece in each of the four main feedwater lines. Each LEFM Check system channel consists of one spool piece and one channel of instrumentation. The LEFM Check system is a digital system controlled by software using the ultrasonic transit time method to measure four velocities at precise locations with respect to the pipe centerline. The system numerically integrates the four measured velocities to determine the mass flow rate and the fluid temperature. These measurements are used by the plant computer to determine the reactor thermal power.

Caldon Engineering Report ER-80P, and its supplement ER-160P, describe the LEFM Check System and provide calculated uncertainties in percent power for a typical pressurized-water reactor (PWR) or boiling-water reactor using measurements by a single meter LEFM Check system. ER-80P and ER-160P provide a generic basis for an uprate up to 1.4% of the licensed reactor power with the use of the LEFM Check System.

ENO stated that the system software had been developed and will be maintained under a verification and validation (V&V) program. Configuration management and software control will be in accordance with IP3's Software Quality Assurance Program. At IP3, the LEFM Check

System is included in the preventive maintenance program. As a plant system, all equipment problems fall under the site work control process. All conditions that are adverse to quality are documented under the corrective action program. Procedures are maintained for notification of deficiencies and error reporting. The IP3 LEFM Check System software is under Caldon's V&V program, and procedures are maintained for user notification of deficiencies that could affect the accuracy and reliability of mass flow and temperature measurements.

The LEFM Check System indications of feedwater flow and temperature will be displayed on a local display panel and transmitted to the plant computer for use in the calorimetric calculation. This information will be directly substituted for the flow nozzle-based flow indications and the thermocouple temperature indications currently used in the plant calorimetric calculation. The flow nozzle-based feedwater flow measurement will continue to be used for feedwater control and other functions that it currently fulfills.

The NRC staff's SER regarding Caldon Engineering Report ER-80P included four additional requirements to be addressed by a licensee referencing ER-80P in their request for a power uprate. ENO's submittals addressed each of the four requirements as follows:

1. The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

    - ENO stated that the procedures for maintenance and calibration of the LEFM Check System were developed for IP3, based on the vendor's recommendations.

      If any of the LEFM Check System channels become inoperable, plant operations may remain at a core thermal power level of 3067.4 MWt for the allowed outage time of 7 days as long as steady-state conditions persist. During the period of outage, the calculation of thermal power will be based on the operable LEFM Check System channels and alternate plant instruments (flow nozzle and thermocouples) in the feedwater line with the out of service LEFM Check system channel. If more than one LEFM Check System channel is out of service, substitute values from the alternate instrumentation will be used for each LEFM Check System channel that is unavailable.

      If the LEFM Check System channels were not returned to operation after 7 days, the power level would be reduced based on the number of the LEFM Check System channels that are available. If all four LEFM Check System channels were unavailable after 7 days, the power level would be reduced to, or maintained at, a power level no greater that 3025 MWt.

      The 7-day allowed outage time is based on trending of the calorimetric of the alternate plant instruments over a fuel cycle and found to be constant during weekly checks. The variation was 0.3% with no consistent drift in any direction. Therefore, if the LEFM Check System fails, the calorimetric would be accurate for the 7-day period.

2. For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Engineering Report ER-80P.

   - ENO stated that an LEFM was originally installed at IP3 in 1982 and the original electronics unit was upgraded to the Caldon LEFM 8300 electronics unit in 1997. An upgrade to the Caldon LEFM Check System electronics unit, which meets the requirements of ER-80P and ER-160P, has been installed at IP3. The installation was performed in accordance with Caldon's installation and commissioning procedures. These procedures were produced in accordance with the descriptions and criteria by the referenced Engineering Report ER-80P. The Caldon LEFM Check System installed at IP3 is representative of the Caldon LEFM Check System discussed in Engineering Report ER-80P, and is bounded by the requirements set forth in this topical report.

3. The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installation for comparison.

   - ENO's submittals included plant specific power measurement uncertainty calculations for IP3. These calculations statistically combined the Caldon provided LEFM Check system feedwater mass flow and temperature measurement uncertainties with other instrumentation uncertainties affecting the plant power calorimetric uncertainty. The resulting calorimetric measurement uncertainty for IP3 was found to be 0.6% of the rated thermal power, which justifies the proposed 1.4% power uprate. The methodology used in these calculations utilizes the square-root-of-the-sum-of-the squares to combine the power measurement uncertainty components. The square-root-of-the-sum-of-the squares methodology is an acceptable methodology for combining instrumentation uncertainties per Instrument Society of America standard ISA-S67.04, Part 1 -1994, "Setpoints for Nuclear Safety-Related Instrumentation." ISA-S-67.04, Part 1 - 1994 is endorsed by NRC RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation."

4. Licensees of plants where the ultrasonic meter (including the LEFM) has not been installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. The justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds (dimensionless index term that defines the type of flow in any pipe) numbers. Additionally, for previously installed calibrated LEFM, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

- ENO stated that the IP3 flow elements were not calibrated in a site specific hydraulic geometry when they were installed in 1982. When upgraded electronics were installed in 1997, Caldon performed feedwater flow measurement uncertainty calculations for IP3. The profile factors assigned to the IP3 flow elements are based on calibration data for similar flow elements at other nuclear power plants, with uncertainty allowances adjusted to reflect these data as well as the hydraulics of the installations. The analysis on which the profile factor selection and its uncertainty are based is contained in Caldon Engineering Report ER-100, as summarized in the ENO's November 20, 2002, submittal.

  The basis for the uprate at IP3 is similar to that for Comanche Peak Units 1 and 2, which was approved by the staff in a safety evaluation report dated March 8, 1999. The IP3 analysis uses a more conservative approach in the selection of a profile factor and in the estimation of the uncertainty associated with the profile factor.

  Recent tests performed by Caldon on LEFM flow elements in straight pipe, with the LEFM at various distances downstream from flow straighteners in a variety of complex hydraulic geometries, provided confirmation of low sensitivity of LEFM flow elements to flow profile. ENO stated that the analysis and tests established a bounding profile factor uncertainty for IP3 that is well within the budgeted allowance for LEFM Check Systems profile factor uncertainty (± 0.4%) in Caldon Engineering Report ER-157N, Revision 5 and ENO's November 20, 2002, submittal. The staff's review of the information in the November 20 submittal confirms this statement. ENO, however, used ± 0.47% uncertainty for the IP3 LEFM Check system profile factor in the plant specific power uncertainty analysis discussed in ENO's response to staff requirement number three. This value of the IP3 profile factor uncertainty is more conservative than the value established by the analysis and, therefore, is acceptable.

The staff finds that ENO's responses have sufficiently resolved the plant-specific concerns regarding LEFM Check System maintenance and calibration, hydraulic configuration, processes and contingencies for an inoperable LEFM Check System, and the methodology for the plant-specific calculations of the IP3 power measurement uncertainty.

Based on the review of ENO's submittals on the LEFM Check system and plant power calorimetric uncertainty, the NRC staff finds that the IP3 thermal power measurement uncertainty with the LEFM Check system is limited to 0.6% of RTP and can support the proposed 1.4% power uprate. The staff also finds that ENO sufficiently addressed the four additional requirements outlined in the staff SER of Caldon Engineering Report ER-80P. The proposed changes to the engineered safety features actuation system (ESFAS) instrumentation allowable values were calculated using an acceptable methodology and, therefore, are acceptable.

3.2 Evaluation of Accident and Transient Analyses

3.2.1 NSSS Design Parameters

In ENO Document No. IP3-RPT-MULT-03614, "1.4% Measurement Uncertainty Recapture Power Uprate Application Report" (Application Report), the licensee provided a list of revised thermal design parameters to reflect the 1.4% increase in the IP3 licensed core thermal power from 3025 MWt to 3068 MWt (conservatively rounded up from 3067.4 MWt). The parameters include reactor power, thermal design flow (TDF), reactor coolant pressure, temperature, and flow rate, feedwater temperature, steam pressure, temperature and flow rate. These key plant parameters have been reconciled with the applicable system and component evaluations, as well as safety analyses, performed in support of the 1.4% power uprate. The set of parameters applicable for NSSS system and component analyses assumes 0% steam generator tube plugging (SGTP) and the set of parameters applicable for accident analyses assumes 24% SGTP, which is consistent with the restriction provided in the current TSs. Since values of these parameters are demonstrated acceptable by various safety analyses using these parameters as inputs, the NRC staff finds that these power uprate parameters are acceptable to support the proposed power uprate.

3.2.2 NSSS Design Transients

As stated in its Application Report, the licensee evaluated the NSSS design transients to account for any impacts from the power uprate. The NSSS design transients are traditionally developed for fatigue analyses of the various NSSS components. Conservatism is generally included in them by means of the analysis assumptions associated with either frequency of occurrences or the transient assumptions. The licensee provided a tabulation, which shows its comparison of operation under the original power rating and 1.4% power uprate conditions. In the list of parameter changes, the licensee has identified that the lower steam temperature associated with the power uprate at full power is not bounded by the existing design transients. To maintain sufficient conservatism in the design transients, the licensee has revised the value of steam temperature in the events associated with load changes using the lower steam temperature at 1.4% power uprate full power conditions. The NRC staff has reviewed the information supporting the licensee's conclusion and finds it acceptable.

3.2.3 Accident and Transient Analyses

The licensee re-evaluated the transient and accident analyses for the operation of IP3 at a rated core thermal power of 3068 MWt with the power measurement uncertainty of 0.6%. The changes to the values of the NSSS parameters discussed in Section 2.1 of the Application Report were used in the analyses to support the power uprate. The licensee performed the majority of the uprate analyses and evaluations in accordance with the current IP3 licensing bases methodologies.

The licensee evaluated the transient and accident analyses described in Chapter 14 of the Final Safety Analysis Report (FSAR) to identify whether each analysis was affected by the 1.4% power uprate. There are 5 categories of events tabulated in Table 8-1 of the Application Report. These are: (1) LOCA-related events, (2) affected non-LOCA events reanalyzed for 1.4% power uprate, (3) affected non-LOCA events evaluated for the 1.4% power uprate using existing departure from nucleate boiling (DNB) margin, (4) non-LOCA events bounded by current 102% power assumption, and (5) non-limiting/bounding events. The NRC staff evaluation of these event categories is addressed below.

3.2.3.1 LOCA-Related Events

Large-Break and Small-Break LOCA (LBLOCA and SBLOCA)

The current licensing basis LBLOCA and SBLOCA analyses employ a nominal core power of 3025 MWt. The current licensing methodology applies a 2% calorimetric power measurement uncertainty allowance resulting in an assumed core power of 3068 MWt in accordance with the original requirements of 10 CFR Part 50, Appendix K. This analytical power level of 3086 MWt is equivalent to the uprated power level of 3067.4 MWt with a 0.6% calorimetric power measurement uncertainty allowance. Therefore, the existing LBLOCA and SBLOCA analyses are still applicable to power uprate conditions at IP3.

Post-LOCA Long-Term Core Cooling

10 CFR 50.46 (b)(5) requires that: “After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.” Since credit for the control rods is not taken for an LBLOCA, the borated ECCS water provided by the refueling water storage tank (RWST) and accumulators must have a boron concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical, assuming all control rods out. The water volumes and associated boron concentration of all the water sources involved following a LOCA are not affected by the 1.4% power uprate. Therefore, the current long-term core cooling analysis of record is unaffected by the 1.4% power uprate.

3.2.3.2 Affected Non-LOCA Events Reanalyzed for 1.4% Power Uprate

Some non-LOCA events are affected by the 1.4% power uprate because the current analyses do not already explicitly account for a 2% power measurement uncertainty allowance. Because of this, the following three events had to be analyzed to address the potential effects of the 1.4% power uprate.

Uncontrolled Control Rod Assembly Withdrawal at Power

To analyze this event, the licensee used the methodology documented in the analysis of record. The methodology utilizes the LOFTRAN computer code for the transient analysis simulation. The Westinghouse reload methodology described in report WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” dated July 1985, is also applied. For the 1.4% power uprate with 0.6% power measurement uncertainty allowance, the calculated minimum DNB ratio (DNBR) is 1.7165, which is higher than the minimum DNBR safety analysis limit of 1.54. The calculated peak primary pressure and peak secondary pressure are 2746 psia and 1204 psia, which are less than 110% of their design pressures (2748.5 psia for primary system and 1208.5 psia for secondary system), respectively. The results of the reanalysis meet the acceptance criteria of Standard Review Plan (SRP) 15.4.8 and, therefore, are acceptable.

Loss of External Electric Load

To analyze this event, the licensee used the methodology documented in the analysis of record. The methodology utilizes the LOFTRAN computer code for the transient analysis simulation. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A is also applied. For the 1.4% power uprate with 0.6% power measurement uncertainty allowance, the calculated minimum DNBR is 2.2343 which is higher than the minimum DNBR safety analysis limit of 1.54. With respect to pressure effects, the turbine trip event is more limiting than any other partial or complete loss of load event, since it results in the most rapid reduction in steam flow. This causes the most limiting increase in pressure and temperature in the primary and secondary systems due to rapid decrease in steam flow. The results of the reanalysis meet the acceptance criteria of this event stated in SRP 15.2.1 and, therefore, are acceptable.

Excessive Heat Removal Due to Feedwater System Malfunctions (Full Power Analysis)

To analyze this event, the licensee used the methodology documented in the analysis of record. The methodology utilizes the LOFTRAN computer code for the transient analysis simulation. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A is also applied. For the 1.4% power uprate with 0.6% power measurement uncertainty allowance, the calculated minimum DNBR is 2.063 for automatic rod control case and 2.113 for manual rod control case which are higher than the minimum DNBR safety analysis limit of 1.54. Since this is a cooldown event, peak pressure is not a concern for this event. The results of the reanalysis meet the acceptance criteria of this event stated in SRP 15.1.1 and, therefore, are acceptable.

3.2.3.3 Affected Non-LOCA Events Evaluated Using Existing DNB Margin

The following non-LOCA events are affected by the 1.4% power uprate, but it was possible to sufficiently address the potential effects through technical evaluation and the use of available DNB margin rather than performing a full analysis.

Rod Assembly Misalignment and Rod Cluster Control Assembly (RCCA) Drop

The dropped RCCA transients were previously analyzed using the methodology described in Westinghouse Report WCAP-1 1394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated October 23, 1989, and examined to determine that the DNB design basis is met. This methodology involves the use of generic statepoints for the dropped rod event. Sensitivity studies on the effect of a power increase on the generic statepoints were previously performed for a four-loop plant. The studies quantified the effect of an approximately 5% power increase on the four-loop generic statepoints. Since the IP3 1.4% power uprate is much smaller than the power increase used in the sensitivity studies, the generic statepoints continue to apply to IP3. The licensee has evaluated the DNB design basis using the generic statepoints and increased nominal heat core flux associated with 1.4% power uprate and confirmed that the DNB design basis continues to be met. Therefore, all applicable acceptance criteria for this event stated in SRP 15.4.3 continue to be met for the 1.4% power uprate.

Loss of Reactor Coolant Flow and Reactor Coolant Pump Shaft Seizure or Shaft Break

Since the 1.4% power uprate could potentially affect the minimum DNBR, an evaluation was completed for these events. The licensee's evaluation concludes that, considering the statepoints affected by the increased nominal heat flux due to 1.4% power uprate, the DNB design limit remains satisfied for these events. With respect to peak system pressures, the current analysis for these events models a 2% power measurement uncertainty, which is applicable for 1.4% power uprate conditions. Therefore, all applicable acceptance criteria for these events stated in SRP 15.3.1 through 15.3.4 continue to be met for the 1.4% power uprate.

3.2.3.4 Non-LOCA Events Bounded by Current 102% Power Assumption

The following non-LOCA events are currently analyzed with an explicit 2% power measurement uncertainty allowance that already bounds operation at the 1.4% uprate power level with the reduced power measurement uncertainty allowance of 0.6%: (1) reactor coolant pump shaft seizure and shaft break (overpressure analysis), (2) loss of external electrical load (overpressure analysis), (3) loss of normal feedwater, (4) loss of all AC power to the station auxiliaries, and (5) rupture of a control rod drive mechanism housing (RCCA ejection). The licensee has evaluated the effects of the small changes in the plant initial operating conditions to these analyses and concluded that the current analyses of record for these events remain valid for the 1.4% power uprate conditions. The staff finds the licensee's assessment acceptable.

3.2.3.5 Non-Limiting/Bounding Events

The following non-LOCA events are either bounded by the current respective analyses of record, or simply are not affected because they are performed starting at hot zero power or a power less than full power: (1) uncontrolled control rod assembly withdrawal from a subcritical condition, (2) chemical and volume control system malfunction, (3) excessive heat removal due to feedwater system malfunctions (zero power analysis), (4) excessive load increase incident, (5) rupture of a steam pipe (zero power analysis), and (6) rupture of a control rod drive mechanism housing (zero power analysis). Since these events are not affected by the rated full power level, the staff considers that the analyses of record remain valid for operating at 1.4% power uprate.

Anticipated Transients Without Scram (ATWS)

IP3 has implemented the ATWS rule, 10 CFR 50.62, by installing a diverse turbine trip and diverse emergency feedwater actuation system. These system designs were approved by the NRC based on their reliability, independence, and diversity from the plant protection system.

The licensee has evaluated the setpoints for these systems and concluded that the setpoints are adequate for protecting against ATWS events and are unchanged for 1.4% power uprate. The licensee indicates that the Westinghouse generic ATWS analysis for four-loop plants is applicable to IP3 with the current plant configuration at a rated power of 3025 MWt except the total auxiliary feedwater pump capacity is 10% less than that assumed in the generic analysis. The peak primary pressure for the limiting loss of load ATWS event in the generic analysis is 2979 psia. The results of the sensitivity studies associated with the generic analysis show that

(1) a 2% increase in power will increase peak pressure by 44 psi and (2) a 50% reduction of emergency feedwater flow will increase peak pressure by 31 psi. Based on the data from the above sensitivity studies, the resulting peak pressure for a limiting loss of load ATWS will remain below the maximum allowable limit of 3200 psia with sufficient margin. Therefore, the NRC staff finds that the design at IP3 regarding ATWS remains effective for a 1.4% power uprate and is acceptable.

3.2.3.6 Event is Prohibited by IP3 TSs

The current IP3 TSs do not allow reactor operation with less than four loops in service. Therefore, an event that involves startup of an inactive reactor coolant loop is prevented and analysis of this event is not required.

3.2.3.7 Evaluation of Radiological Consequences

The NRC staff reviewed the impact of the proposed changes on design-basis accident (DBA) radiological analyses, as documented in Chapter 14 of the IP3 FSAR. In its submittal, the licensee identified the existing DBA radiological analyses of record by their location in the IP3 FSAR. The licensee stated that the current radiological analyses of record for IP3 were unaffected by the requested power uprate, because they were performed assuming either a nominal core power of 3025 MWt with a 2% uncertainty allowance or a core power of 3216 MWt, both of which bound the conditions for the requested 1.4% power uprate. Using the current IP3 FSAR documentation in addition to information in the May 30, 2002, submittal letter, the staff verified that the existing IP3 FSAR Chapter 14 radiological analyses source term and steam release assumptions, as appropriate, bound the proposed 1.4% power uprate conditions.

Based on the above discussion, the staff finds that the existing IP3 FSAR Chapter 14 radiological analyses, which were analyzed at 102% or more of the current rated core thermal power of 3025 MWt, remain bounding for the proposed 1.4% power uprate to 3067.4 MWt, considering the higher accuracy of the Caldon LEFM flow meters. These analyses of record show that the radiological consequences of postulated DBAs meet the dose limits given in 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, as well as applicable dose acceptance criteria given in SRP Chapter 15. Therefore, the staff finds that the proposed changes are acceptable with respect to the radiological consequences of DBAs.

3.2.3.8 Summary

The NRC staff has reviewed the licensee's analyses and proposed TS changes to support operations of the IP3 at uprated power level of 3068 MWt. Based on this review, the staff finds that the supporting safety analyses are performed with the previously NRC-approved methods; the input parameters of the analysis adequately represent the plant conditions at the uprated power level; and the analytical results are within the applicable acceptance criteria. Therefore, the staff concludes that the supporting analyses are acceptable.

3.3 Mechanical, Structural, and Material Component Integrity and Design

3.3.1 Nuclear Fuel

The licensee has performed evaluations to determine the potential effects of the 1.4% power uprate on the nuclear fuel at IP3. Fuel evaluations are performed for each specific cycle according to the Westinghouse Reload Methodology in WCAP-9272-P-A. The evaluation herein addresses fuel-related analyses that are not cycle-specific. This evaluation was done based on both Westinghouse VANTAGE+fuel (currently being used in the Cycle 12 core at IP3) and VANTAGE5 fuel (potentially used in future cycle cores at IP3). The design margin for fuel rod internal pressure and cladding stress was re-evaluated based on the 1.4% power uprate conditions. The results indicate that these fuel rod design parameters continue to meet the acceptance criteria at the 1.4% power uprate conditions. Therefore, the NRC staff finds the fuel design acceptable to support the proposed power uprate.

3.3.2 Core Thermal-Hydraulic Design

The licensee has performed a thermal-hydraulic evaluation at the 1.4% power uprate conditions. The evaluation was based on the 15x15 VANTAGE+ fuel design with the intermediate flow mixing grids that bounds the 15X15 VANTAGE5 fuel design for future reloads. The current design methodology for the IP3 reload safety evaluation remains unchanged for the 1.4% power uprate evaluation. The WRB-1 DNB correlation and the Westinghouse revised thermal design procedure DNB methodology are continuously used for DNB analysis. The W-3 DNB correlation is used for events where the conditions fall outside the applicable range of the WRB-1 correlation. The licensee’s evaluation concludes that the current core operating limits and the DNB limiting events continue to meet the DNB design basis at the 1.4% power uprate nominal core power level of 3067.4 MWt. Based on the above stated evaluation results, the NRC staff finds the core thermal-hydraulic design acceptable to support the proposed power uprate.

3.3.3 Reactor Pressure Vessel

The licensee evaluated the RPV for the effects of the revised design parameters in Table 2-1 of the Application Report on the most limiting vessel locations with regard to ranges of stress intensity and cumulative fatigue usage factors (CUFs) in each of the components, as identified in the original reactor vessel stress reports. The evaluations considered the operating parameters, which were identified for the uprated power condition. The existing NSSS design transients were not affected by the 1.4% uprating. The components of the reactor vessel affected by the power uprate include outlet nozzles, the RPV (main closure head flange, studs, and vessel flange), and CRDM housing. The licensee evaluated the maximum stresses and CUFs for the critical components at the core power uprated conditions. The evaluation was performed in accordance with the ASME Code, Section III, 1965 Edition with addenda through Winter 1965, which is the code of record. There are no changes for the faulted condition loads as a result of the 1.4% power uprate. The results for the faulted condition were previously evaluated in accordance with Appendix F of the 1974 Edition.

The calculated maximum stresses and the maximum CUFs for the reactor vessel critical locations are provided in Application Report. The results indicate that the maximum stresses are within the allowable limits, and the CUFs remain below the allowable ASME Code limit of 1.0. The licensee concluded that the current design of the reactor vessel continues to be in compliance with licensing basis codes and standards for the power uprate condition. Based on its review, the staff agrees with the licensee's conclusion. On the basis of its review of this information, the NRC finds this acceptable.

Regarding the IP3 RPV surveillance program and capsule withdrawal schedule, the licensee stated in its Application Report Sections 7.2.2 and 7.2.3:

> The current surveillance capsule withdrawal schedule for IP3 is documented in Westinghouse Report WCAP-11057, Revision 1, "Indian Point Unit 3 Reactor Vessel Fluence and RT-PTS Evaluations for Consideration of Life Extension," dated June 1989. As part of the 1.4% power uprate, the capsule fluence values only changed slightly from those documented in WCAP-11057. Therefore, a new surveillance capsule withdrawal schedule was calculated based on ASTM E185-82 [Standard Practice for Conducting Surveillance Tests For Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)]. Per ASTM E185-82, the withdrawal of a capsule is to be scheduled at the nearest vessel refueling outage to the calculated EFPY established for the particular surveillance capsule withdrawal.
>
> Therefore, the capsules removed from the IP3 vessel to date meet the intent of ASTM E185-82. Since the capsule fluences for the 1.4% power uprate have changed from the capsule fluences used in development of the current withdrawal schedule, a new withdrawal schedule was developed and documented in Table 7-1. Table 7-2 shows that the maximum, ΔRT_{NDT} using the 1.4% power uprate fluences for IP3 at 27.1 EFPY is 217.1°F. Per ASTM E185-82, this, ΔRT_{NDT} value would require five capsules to be withdrawn from IP3.

Based on the revised RPV fluence information submitted in the licensee's May 30 application and the staff's review of the current IP3 surveillance capsule withdrawal schedule, the NRC staff finds the licensee's conclusion acceptable since the requirements of Appendix H to 10 CFR Part 50 will continue to be met.

3.3.4 RPV Fluence

Fluence Calculational Methodology

Fluence projections on the RPV were calculated for the 1.4% power uprate for the evaluation of the PT curves and the end-of-license RT_{PTS}. Information provided in the Application Report demonstrates that the cross section data, the cross section approximations, the geometrical approximations, and the analytical formulations conform to the guidance of RG 1.190. The calculated fluences used in the 1.4% power uprate conform to the guidance in RG 1.190 and, therefore, are acceptable.

Pressure-Temperature (P-T) Curves

The licensee proposed to lower the period of applicability of the P-T curves from 16.20 effective full-power years (EFPYs) of operation to 16.17 EFPYs. At the time of the planned implementation of the power uprate, the reactor will have accumulated 15.30 EFPYs. The 0.90 difference in EFPYs will be at a 1.4% higher power level. The excess EFPY will be 0.90 x 0.014 = 0.0126 and the Δt at the 1.014 power level will be $\Delta t = 0.0126/1.014 = 0.0124$ EFPY. The revised operating time will thus be 16.188 EFPYs. Therefore, since the proposed value of 16.17 EFPY is conservative, the NRC staff finds it acceptable.

In Sections 7.2.2 and 7.2.3 of its Application Report regarding the topic of the RPV P-T limits, the licensee stated that:

> Review existing Pressure-Temperature (P-T) limit curves to determine if a new applicability date needs to be calculated due to the effects of the 1.4% power uprate fluence projects. The methodology of NRC Regulatory Guide 1.99, Revision 2 (May 1988) *[Radiation Embrittlement of Reactor Vessels Materials*], will be used in any required calculations.
>
> The applicability date for the heatup and cooldown curves presently identified by IP3 Tech Spec. 3.4.3 shall be known.
>
> IP3 Tech Spec 3.4.3 shows the current applicability date of 16.2 EFPY. Because of the 1.4% power uprate fluences, the applicability date has been decreased slightly to 16.17 EFPY as shown in Table 7-3. As shown, this change equates to less then one month of operating time and is based on current Accumulated Lifetime Burnup, the time remaining on the current P-T curves and the 1.4% power uprate.

In addition, the NRC staff reviewed related information regarding the licensee's updated RPV fluence information. Based on the NRC staff's approval of the current IP3 P-T limit curves for 16.2 EFPY and the aforementioned updated RPV fluence information, the NRC staff concluded that the existing IP3 P-T limit curves will continue to be acceptable, based on meeting the requirements of Appendix G to 10 CFR Part 50, for operation up to at least 16.17 EFPY after the requested power uprate is implemented. In fact, the NRC staff noted that this action, to reduce the period of applicability of the IP3 limit curves, was conservative given the overall projected RPV fluence values, which were cited in the licensee's May 30 application.

Pressurized Thermal Shock (PTS) Nil Ductility Reference Temperature (RT_{PTS})

As discussed above, the licensee used acceptable methods for the calculation of the projected fluence value to the end-of-life (EOL). The 1.4% power uprate RT_{PTS} values for all beltline materials must not exceed the screening criteria as specified in 10 CFR 50.61. Specifically, the RT_{PTS} values of the base metal (plates or forgings) shall not exceed 270°F, while the girth weld metal RT_{PTS} values shall not exceed 300°F through the EOL.

The licensee presented the results of its calculations for the IP3 beltine region materials at EOL in Table 7-5 of the Application Report. Using the 1.4% power uprate fluences, the most limiting beltline material in the IP3 reactor vessel remains below the screening criteria values of 270°F at EOL (27.1 EFPY). The critical material in the beltline region is the lower shell plate with an RT_{PTS} = 263 °F, which is within the limits of the 10 CFR 50.61 screening criterion; therefore, the RT_{PTS} is acceptable.

The RT_{PTS} value of record was 264.6 °F for an EOL of 27.1 EFPYs. In its Application Report, the licensee clarified that a very low leakage program has been implemented to lower the EOL fluence in peak azimuthal locations. As described in FSAR Section 3.2, hafnium flux suppressors have been placed in the eight core locations nearest to the vessel to prolong vessel life.

Upper Shelf Energy (USE)

With USE analyses for the IP3 RPV, the licensee stated that the EOL USE values for all reactor beltline materials meet the requirements of 10 CFR 50, Appendix G, in that all beltline materials are expected to have a USE greater than 50 ft-lb through EOL. The licensee also stated that the EOL (27.1 EFPY) USE was predicted using the EOL 1/4T fluence projection and the predictions are shown in Table 7-6 of the Application Report.

In its letter dated November 6, 2002, the licensee provided additional information regarding USE calculations for the Beltline Region Materials contained in Table 7-6:

> ENO has confirmed that an error was made in recording the ‘projected upper shelf energy (USE) decrease’ values from Regulatory Guide 1.99. The corresponding projected end-of-life (EOL) USE values are therefore also incorrect. A corrected page for Table 7-6 is provided at the end of this attachment. The limiting value for projected EOL USE is for the lower shell plate B2803-3 and is based on surveillance capsule data. The corrected Table also includes this clarification. These corrections do not alter the conclusion in the original submittal that the requirements of 10 CFR 50 Appendix G are met for the proposed power uprate.

The NRC staff has evaluated the information provided by the licensee as well as information contained in the staff’s Reactor Vessel Integrity Database. Based on the revised fluence values noted in the corrected Table 7-6, the NRC staff independently confirmed that the IP3 RPV materials would continue to meet the USE requirements of Appendix G to 10 CFR Part 50 through EOL.

3.3.5 Reactor Core Support Structures and Vessel Internals

The licensee indicated that 1.4% power uprate does not affect the current design basis seismic, LOCA loads and the design basis NSSS transients. The evaluation of the reactor vessel core support and internal structures was performed for the power uprate condition. The limiting reactor internal components evaluated include the lower core plate, the baffle barrel region components (including core barrel, baffle plate, baffle/barrel region bolts), and the upper core plate. In its November 6 response to the staff’s request for additional information, the licensee indicated that the reactor internal components were not licensed to the ASME Code. However, the design of the IP3 reactor internals was evaluated in accordance with requirements of Subsection NG of the 1986 Edition of the ASME Code Section III. The NRC finds this acceptable.

The licensee evaluated critical reactor internal components considering the revised design conditions provided in Table 2-1 of the Application Report. The licensee indicated that for the baffle-barrel region components, the current structural and thermal analyses of record for IP3 remain bounding for the proposed power uprate condition. Table 7-7 of the Application Report provides the maximum calculated stress intensity and CUF for the lower and upper core plates. The calculated stresses are shown to be less than the allowable stress limits and the CUFs are less than the limit of 1.0. The remaining reactor internal components are less limiting. In addition, in Section 7.3 of the Application Report, the licensee concluded that the IP3 analysis of record for the reactor internals indicated that the reactor internals hold-down forces are not affected by the increased flow as a result of the 1.4% power uprate. Based on these evaluations, the licensee concluded that the reactor internal components at IP3 will be structurally adequate for the proposed power uprate. The staff finds the licensee's assessment to be acceptable.

In addition, the licensee stated in Section 7.3.3 of the Application Report:

> Evaluations were performed to demonstrate that structural integrity of the reactor internal components is not adversely affected by the 1.4% power uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which are accounted for in the design and analysis of various components.
>
> The core support structure components affected by the 1.4% power uprate are discussed below. The primary inputs to the evaluations are the revised RCS temperatures (as discussed in Section 2) and the gamma heating rates. The gamma heating rates took into account the 1.4% increase in core power.
>
> The reactor internals components subjected to heat generation effects (either directly or indirectly) are the upper core plate, the lower core plate, and the baffle-barrel region. For all of the reactor internal components, except the lower core plate and the upper core plate, the stresses and CUF factors were unaffected by the 1.4% power uprate conditions because the previous analyses remain bounding.

Based on the information provided by the licensee regarding changes to operating temperature, flow rates, and neutron fluences that result from the power uprate, the NRC staff agrees that the integrity of the RPV internals will be maintained such that the licensee's ability to meet the regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry will not be adversely impacted.

3.3.6 Control Rod Drive Mechanisms (CRDMs)

The pressure boundary portion of the CRDMs is that portion exposed to the vessel/core inlet fluid. The licensee evaluated the adequacy of the CRDMs by reviewing the IP3 current CRDM design specifications and stress report to compare the design-basis input parameters against the revised design conditions in Table 2-1 of the Application Report. The comparison shows that the current design analyses are bounding for the 1.4% power uprate. Therefore, the existing stress and fatigue usage are not affected for the power uprate and continue to satisfy the allowable stress and fatigue usage limits.

In Section 7.5 of the Application Report, the licensee also concluded:

> The CRDM parameters are based on the hot leg data, which is the vessel outlet temperature data presented in Table 2-1. The evaluation was performed for the 1.4% power uprate NSSS power of 3082 MWt (3067.4 MWt core power). The upper bound vessel outlet temperature is shown to decrease from the current design operating temperature of 601.5 °F to 600.8 °F. The higher temperature evaluated for the current analysis remains bounding for the 1.4% power uprate. The design temperature and pressure for the CRDMs are 650 °F and 2485 psig, respectively.
>
> The revised pressure changes (ΔPs) and temperature changes (ΔTs) of the transients are less than those previously evaluated and continue to bound the 1.4% power uprate, with the exception of the Unit Loading (Unloading) transient applied to the CRDM middle joint cono-seal hold down sleeve. The original design transient analysis for this location considered an alternating stress range of:
>
> $$S=½\cdot K_f\cdot E\cdot\alpha\cdot\Delta T$$
>
> where: K_f = stress concentration factor, taken as 4,
>
> E = modulus of elasticity, taken as 25.4 x 10^6 psi for austenitic stainless steel at a temperature of 600°F, and
>
> α = mean coefficient of thermal expansion, taken as 9.82 x10^{-6} inches/inch/°F for austenitic stainless steel between temperatures of 70°F and 600°F, and
>
> ΔT = temperature difference, taken as 25 °F in the original analysis.
>
> The required ΔT is 65.6°F for the Unit Loading/Unloading transient. This difference in temperature results in an increase in the cumulative usage factor from 0.066 to 0.118, using the design fatigue curve in the Code of Record, which is the 1965 Edition of the ASME Code, Section III, with Addenda through Summer 1966. This CUF factor remains well below the allowable limit of 1.0. The acceptability of the CRDMs is thus demonstrated for the 1.4% power uprate conditions.

Based on the information provided by the licensee regarding the CRDMs, the NRC staff finds that the licensee demonstrated the acceptability of the CRDMs for the 1.4% power uprate conditions. Therefore, the integrity of the CRDMs will be maintained such that the licensee will be able to meet the regulatory requirements in Appendix A to 10 CFR Part 50 and GDC-14. On the basis of its review, the staff finds acceptable the licensee's conclusion that the current design of CRDMs continues to be in compliance with the licensing basis codes for the power uprate conditions.

3.3.7 Reactor Coolant Pumps (RCPs)

The licensee reviewed the existing design basis analyses of the IP3 RCPs to determine the impact of the revised design conditions in Table 2-1. The licensee indicated that the applicable ASME Codes used in the evaluation are the same as the Code of record.

After the core power uprate, the RCS pressure remains unchanged. The most limiting design parameter of the steam generator (SG) outlet temperature, as provided in Table 2-1 of the Application Report, was decreased for the power uprate condition, in comparison to the design temperature defined in the RCP equipment specification. Typically, a higher SG outlet temperature results in a greater actual stress. As a result of the evaluation, the licensee indicated that the current stress and CUFs in the stress reports for the IP3 RCPs remain bounding for the 1.4% power uprate.

On the basis of its review, the staff concurs with the licensee's conclusion that the current RCPs, when operating at the proposed conditions with a 1.4% power increase from the current rated power, will remain in compliance with the requirements of the codes and standards under which IP3 was originally licensed.

3.3.8 Pressurizer

The licensee evaluated the structural adequacy of the pressurizer and components for limiting locations at the pressurizer spray nozzle, the surge nozzle, and upper shell for operation at the uprated conditions. The evaluation was performed by comparing the key parameters in the current IP3 pressurizer stress report against the revised design conditions in Table 2-1 for the proposed power uprate. Section 7.8 of the Application Report provides the comparison of the current and uprated pressurizer design parameters. For components affected by the hot leg temperature (e.g., surge line), the temperature difference between the pressurizer and the hot leg is 52.2 °F for the power uprate. This is bounded by the temperature difference at the current operating condition. The decrease in temperature difference represents a reduction of the associated thermal stress in the components. The limiting component affected by the cold leg temperature is the spray nozzle, for which the temperature difference between the pressurizer and the cold leg is 111.1 °F, which is bounded by the design basis temperature difference of 125 °F. Therefore, the existing design basis analyses remain valid for the proposed power uprate. The licensee concluded that the existing pressurizer components will remain adequate for plant operation with the proposed 1.4% power increase while the RCS pressure remains unchanged. Based on its review, the staff agrees with the licensee's conclusion.

3.3.9 NSSS Piping and Pipe Supports

The proposed power uprate of IP3 involves the increase of temperature difference across the RCS. The licensee evaluated the NSSS piping and supports by reviewing the existing design basis analysis against the uprated power condition, with regard to the design system parameters, transients and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. The methods, criteria and requirements used in the existing design basis analysis for IP3 were used for the power uprate evaluation.

The RCS pressure remains unchanged for the proposed core power uprate. The actual hot leg temperature for the power uprate is projected to be slightly greater than the hot leg temperature at the current rated power level. The cold leg temperature for the power uprate condition will be less than the cold leg temperature at the current power level. The licensee indicated that the thermal expansion analysis performed for the RCL piping and the pressurizer surge line envelopes the revised design parameters as identified in Table 2-1. Therefore, the current results for the RCL piping and the pressurizer surge line remain bounding and applicable for the 1.4% power uprate.

The licensee also indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for the proposed power uprate. The potential for a slight increase in loop hydraulic forces due to the decrease in the cold leg temperature and the increase in water density at the power uprate condition was offset by the existing margin in the current design basis analysis. The RCL piping evaluation was performed based on United States of America Standard (USAS) B31.1, “Power Piping,” 1967 Code, which is the Code of record. As a result of its evaluation, the licensee concluded that the existing stresses and loads remain bounding for the power uprate for the NSSS components including the reactor cooling loop piping, the primary equipment nozzles, the primary equipment supports, pipe supports and the auxiliary equipment (i.e. heat exchangers, pumps, valves and tanks).

On the basis of its review of the licensee’s submittal, the staff concurs with the licensee’s conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the design bases criteria, as defined in the IP3 FSAR and are, therefore, acceptable for the proposed power uprate.

<u>Leak Before Break Methodology</u>

In Section 7.4.3 of the Application Report, the licensee concluded that:

> There is no significant change in loads due to the 1.4% power uprate parameters as indicated in Section 7.4. 1. The effect of material properties due to the changes in temperature, shown in Table 2-1, will have a negligible effect on the LBB margins. The existing LBB analysis conclusions remain applicable for the 1.4% power uprate conditions for IP3.
>
> Therefore, based on the above assessment, the LBB margins will be negligibly affected, and the conclusions of Westinghouse Report WCAP-8228, Volume 1, Revision 1, "Structural Evaluation of the Reactor Coolant Loop/Support System for Indian Point Nuclear Generation Station Unit No. 3," dated April 1997, for LBB remain unchanged and applicable for the 1.4% power uprate conditions for IP3.

Based on the changes in pressure, temperature, and operating loads expected to result from the proposed 1.4% power uprate, the NRC staff agrees with the licensee’s conclusion that the effect of the proposed power uprate on the facility’s existing LBB evaluations will be insignificant. However, due to recent events concerning primary water stress corrosion cracking (PWSCC) of Inconel 82/182 material, the staff is in the process of examining the significance of this issue with respect to existing LBB evaluations. Currently, the staff is evaluating what licensee actions, if any, are necessary to ensure that the technical bases for existing LBB approvals remain valid, and any concerns regarding the effect of PWSCC on existing LBB evaluations will be resolved separately from this power uprate submittal. As noted above, the staff also expects the impact of the proposed power uprate, as well as the small projected increase in T_{hot}, on the PWSCC susceptibility of any Alloy 82/182 materials in lines approved for LBB at IP3, to be insignificant.

Summary

The staff has reviewed information provided in Sections 7.2.2, 7.2.3, 7.3.3, 7.4.3, 7.5 and Table 7-6, Revision 1, of the Application Report, and in the November 6, 2002, letter, and the proposed modifications to the IP3 TS Figures 3.4.3-1 and 3.4.3-2. The staff has concluded that sufficient information regarding the continued acceptability, with respect to the applicable NRC regulations, of the IP3 RPV, RPV internals, and LBB-approved piping has been provided to support NRC approval of a 1.4% power uprate for this unit. Further, the staff finds the revised P-T limit curves (TS Figures 3.4.3-1 and 3.4.3-2) to be acceptable for up to 16.17 EFPY of operation for IP3.

3.3.10 Steam Generators

The licensee reviewed the existing structural and fatigue analyses of the Model 44F SGs at IP3. The revised design conditions for the 0% SGTP condition, contained in Table 2-1, for the power uprate are compared against the design parameters in the Model 44F SG stress reports. Based on the comparison of key input parameters, the licensee developed scaling factors, which were used to scale up the original stress and fatigue usage for the power uprate conditions. The evaluation was performed in accordance with the requirements of the ASME Code, Section III, 1965 Edition through the Summer 1966 Addenda, which is the Code of record for SGs at IP3.

The calculated maximum stresses and cumulative fatigue usage factors for the critical SG components are provided in Table 7-9 of the Application Report. The results indicate that the maximum calculated stresses are below the Code-allowable limits. The results provided in Table 7-9 also show that the calculated CUFs are within the allowable limit of unity. In its November 6 response to the staff’s request for additional information, the licensee provided its evaluation of the flow induced vibration for U-Bend tubes due to the increased feedwater flow rate. The evaluation showed that the maximum fluid-elastic stability ratio and the maximum vibration induced displacement are within the allowable limits and, therefore, acceptable for the proposed power uprate.

On the basis of its review, the staff concludes that the licensee has demonstrated the maximum stresses and CUFs for the critical SG components are within the Code allowable limits and, therefore, acceptable for the proposed 1.4% power uprate.

Structural Integrity Evaluation

The licensee performed a structural integrity evaluation for the SGs, which focused on primary-side components, secondary-side components, potential tube damage (i.e., undercuts) and repair hardware. The main primary-side components that were assessed are the divider plate, tubesheet and shell junctions, tube-to-tubesheet weld, and the SG tubes. The main secondary-side components that were assessed are the feedwater nozzle and secondary manway studs. In addition, mechanical plugs, shop welded plugs and tube undercuts were evaluated.

Comparisons of the primary-side transients and RCS parameters from the current versus power uprated conditions were performed to determine the scale factors that would be applied to the baseline analyses maximum stress ranges and fatigue usage factors. The baseline analysis results for various components were then updated for the 1.4% power uprate conditions.

The 1.4% power uprate structural evaluation was performed for the 3082 MWt NSSS power and 0% SG SGTP condition, since the plant is currently at 0% SGTP. However, for the purposes of IP3 FSAR Chapter 14 safety analyses, the plant has been analyzed to a maximum of 24% SGTP, which results in more challenging operating parameters (i.e., higher pressure differential across the SG tubes). The licensee committed to update the FSAR during the next annual update to document that the structural evaluations have only been performed for the 0% SGTP condition. This ensures that if the licensee were to increase the amount of SGTP at IP3, a 10 CFR 50.59 evaluation would be required, the appropriate structural integrity analysis would be performed, and the NRC would be notified, if required by 10 CFR 50.59.

The licensee compared the results of the structural evaluations against ASME Code, Section III acceptance criteria and concluded the following:

> All components analyzed meet ASME Code, Section III limits. The design pressure requirements of the ASME Code continue to be satisfied. The mechanical plug designs satisfy all applicable stress and retention acceptance criteria. The shop welded plug design satisfies all ASME Code, Section III allowable values. Lastly, the stress evaluation indicated that stresses and fatigue usage factors due to tube end undercut, if needed in support of tube plugging and sleeving activities, would be within ASME Code allowable values under the power uprate conditions.

The NRC staff finds the licensee's evaluation and reasoning to be acceptable and, therefore, the staff concludes that the proposed 1.4% power uprate for IP3 will not have significant impact on SG structural integrity.

<u>SG Tube Wear and Other Modes of Tube Degradation</u>

The potential effects of the 1.4% power uprate on SG tube wear and other modes of SG tube degradation (e.g., axial and/or circumferential cracking, etc.) were evaluated.

The evaluation of the effects of the 1.4% power uprate on SG tube wear was based on the current design basis analysis and consideration of the changes in the thermal-hydraulic properties of the secondary side of the SG resulting from the power uprate. The licensee determined that the increase in the wear rate for the SG tubes was calculated to be 4%, which would result in an estimated tube wear of 0.0013-inch over a 40-year operating period. The amount of wear is lower than the tube wear allowance used to determine the minimum tube wall thickness. The licensee concluded that the increase in the wear rate resulting from the 1.4% power uprate will not result in unacceptable wear.

The licensee evaluated the effects of the power uprate on modes of SG tube degradation other then wear. The licensee does not expect the 1.4% power uprate to have a significant impact on corrosive SG tube degradation. This is based on: the expected corrosion resistance associated with the Alloy 690 thermally treated (TT) SG tubing; IP3's low operating temperature (as compared to other PWRs with Alloy 690 TT tubing); the relatively small increase in the operating temperature at IP3 due to the 1.4% power uprate; and industry experience with Alloy 690 TT and Alloy 600 TT tubing.

Because none of the potential degradation mechanisms are significantly affected by the 1.4% power uprate conditions, the licensee concluded that the required frequency of inspection is also not affected significantly by the planned power uprate.

The NRC staff finds the licensee's evaluation and reasoning to be acceptable.

Secondary Side Foreign Object Evaluation

The licensee indicated that foreign object search and retrieval operations performed during previous refueling outages at IP3 have determined that a number of un-retrievable objects may remain present inside the steam generators. Analyses were performed at that time to determine the amount of time required for the limiting foreign object to wear a tube down to a minimum allowable tube wall thickness. These analyses were assessed to determine the potential effects of the 1.4% power uprate on the projected wear times of the objects. The licensee concluded that the changes in the previously calculated secondary side foreign object wear times are not significant and do not affect the results of the previous analyses. The NRC staff finds the licensee's evaluation and reasoning to be acceptable.

Regulatory Guide 1.121 Analysis

NRC RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection must be removed from service. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating, into the resulting structural limit, a growth allowance for continued operation and an allowance for eddy current measurement uncertainty.

The licensee performed an analysis to define the structural limits for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. In addition, the licensee used ASME Code minimum material properties, which it determined provides additional conservatism. The licensee factored in information based on predicted flaw growth rates and non-destructive evaluation uncertainties and concluded that the current TS repair limit of 40% through wall is adequate. The NRC staff finds the licensee's evaluation and reasoning to be acceptable because it follows the guidance in RG 1.121.

3.3.11 Flow-Accelerated Corrosion

Flow accelerated corrosion (FAC) is a corrosion mechanism causing wall thinning of high energy pipes in the power conversion system which may lead to their failure. Since failure of these pipes may result in undesirable challenges to the plant's safety systems, the licensee has a program for predicting, inspecting, and repairing or replacing the components whose wall thinning exceeds the values required for their safe operation. The program uses the Electric Power Research Institute (EPRI) developed CHECWORKS computer code for predicting thinning of the walls in the components subjected to FAC. In the submittal, the licensee stated it would revise the code to incorporate flow and process system conditions as are determined for the 1.4% power uprate conditions, and that the results of the upgraded code would be factored into future surveillance/pipe repair plans. The staff considers the licensee's action adequate for ensuring integrity of the high energy pipes.

3.4 Electrical Equipment Design

The main generator is rated at 1125.6 MVA (based on 75 psig hydrogen pressure) at a 0.9 power factor (pf). The station output generated at 22 kV, is fed through an isolated phase bus to the primary windings of the two main transformers where it is stepped up to 345 kV voltage and delivered to the switchyard. The unit auxiliary transformer (UAT) supplies power to the balance of plant (BOP) systems under normal operating conditions. The startup transformers provide power to Class 1E and BOP systems under abnormal operating conditions. The station distribution system consists of various auxiliary electrical systems to provide electrical power during all modes of operation and shutdown conditions. The electrical distribution system has been previously evaluated to conform to GDC-17. The plant has also been previously evaluated for environmental qualification for electrical equipment pursuant to 10 CFR 50.49, and Station Blackout pursuant to 10 CFR 50.63.

3.4.1 Emergency Diesel Generators (EDGs)

There is no change to the safety-related loads at uprate conditions and, therefore, the EDGs will not be affected by power uprate and can perform their safety-related functions during a loss of offsite power (LOOP)/LOCA.

The staff's review determined that the power uprate does not affect the loading on the EDGs. Therefore, the licensee will continue to meet GDC-17 requirements with the power uprate.

3.4.2 Station Blackout (SBO) Equipment

Station Blackout is defined in 10 CFR 50.2 as the complete loss of preferred offsite and Class 1E onsite emergency ac power system. The licensee's analysis indicated that the condensate inventory requirements are bounded by the current SBO analysis. There is also no change in the ability of the turbine-driven auxiliary feedwater pump, supplied with steam from the SGs, to support reactor heat removal due to the 1.4% power uprate. In addition, the Appendix R diesel generator is available to provide additional AC power as part of an enhanced alternate shutdown capability. All other associated parameters are either independent of the core power level, or have margin available to accommodate the 1.4% power uprate. The methodology and the assumptions associated with the SBO analysis with regard to equipment operability are unchanged with uprate. Therefore, the ability of the plant to respond to an SBO will not be altered due to the power uprate.

The NRC staff has concluded that the uprate will not adversely affect the ability of the plant to mitigate a postulated SBO event for the uprate condition and the plant will continue to meet the requirements of 10 CFR 50.63 and the design is, therefore, acceptable.

An SBO does not assume the loss of available AC power to buses fed by the station batteries through inverters. IP3 station batteries have been sized to carry their expected shutdown load following a plant trip and a loss of AC power for a period of 2 hours. The alternate AC source is available within 1 hour. The IP3 SBO coping period is 8 hours prior to restoration of an AC power source. Since systems associated with SBO are not affected by the 1.4% power uprate, the turbine driven emergency feedwater (EFW) pump will be available for decay heat removal following the SBO. Since a motor driven EFW pump with a design capacity of 400 gallons per minute (gpm) is sufficient for decay heat removal during a loss normal feedwater transient, the turbine driven EFW pump with 800 gpm capacity is more than sufficient to remove decay heat following an SBO at 1.4% power uprated conditions at IP3. The staff has reviewed the licensee's submittal and concluded that the SBO at IP3 is not affected by the 1.4% power uprate.

The methodology and the assumptions associated with the SBO analysis with regard to equipment operability are unchanged with uprate. There is no change in the ability of the turbine driven auxiliary feedwater pumps, supplied with steam from the SGs, to support reactor heat removal due to the 1.4% power uprate. Therefore, the ability of the plant to respond to an SBO will not be altered due to the power uprate.

The staff reviewed the licensee's submittal and concluded that the uprate will not adversely affect the ability of the plant to mitigate a postulated SBO event and the plant will continue to meet the requirements of 10 CFR 50.63 and the design is, therefore, acceptable.

3.4.3 Environmental Qualification of Electrical Equipment

In accordance with 10 CFR 50.49, electrical equipment that is relied upon to remain functional during and following design basis events must be qualified to survive the environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence, at their specific location during normal and accident operating conditions. The licensee analyzed the impact of a 1.4% power uprate on the normal design temperatures and the environmental conditions in the containment and the primary auxiliary building. The 1.4% power uprate will increase the activity level in the core by the percentage of the core uprate. The radiation source terms in equipment/structures containing post accident fluids, and the corresponding post-LOCA dose rates in the plant will also increase by the percentage of the power uprate. The post-accident environmental dose estimates in the containment are based on a LOCA and a power level of 3280 MWt, which encompasses operation at the 1.4% power uprate conditions. The licensee determined that the current accident analysis bounds the 1.4% power uprate conditions and that the analysis of record regarding equipment qualification (EQ) for harsh environment remains valid for the power uprate. The licensee also determined that the 1.4% power uprate would not significantly alter any normal or abnormal operating conditions or environmental evaluations which would affect equipment qualification.

In addition, the licensee stated that a comparison of the component "specifications" versus "qualification" doses associated with the safety-related components in the IP3 EQ program indicated that there is sufficient available margin to accommodate a 1.4% power uprate.

The 1.4% power uprate will not affect seismic or other natural phenomena aspects of the plant design and, therefore, will have no effect on the IP3 Seismic Qualification or Individual Plant Evaluation for External Effects (IPEEE) program requirements.

Based on its evaluation, the staff concluded that there is sufficient available margin to accommodate a 1.4% power uprate and the plant would meet the requirements of 10 CFR 50.49 and the margin requirements of IEEE 323 at the power uprated conditions.

3.4.4 Grid Stability

The licensee performed the grid stability study in January 2001. The current operating point of the main generator is 1027 MWe at 0.9126 power factor (pf). At the 1.4% power uprate, the main generator will operate at 1041 MWe at 0.9254 pf. The generator capability curve shows that the operation at power uprated condition is possible at a power factor of 0.9254. The main generator can operate within the range of 427.6 megavolt-amphere reactive (MVAR) lagging and 170 MVAR leading. The leading limit is a result of instability limits shown on the generator capability curve. The study identified no stability issues for the 1.4% measurement uncertainty recapture uprate for the unit.

The staff reviewed the licensee's submittal and concluded that, since the generator rating assumed in the grid stability study substantially bounds the proposed power uprate of 1.4%, the plant will continue to meet GDC-17 for grid stability with this power uprate.

3.4.5 Plant Equipment

<u>Main Generator</u>

The main generator is rated at 1125.6 megavolt-amphere (MVA) (based on 75 pounds per square inch guage (psig) hydrogen pressure) at a 0.9 power factor (pf). The current operating point of the main generator is 1027 MWe at 0.9126 pf. At the 1.4% power uprate, the main generator will operate at 1041 MWe at 0.9254 pf (1124.9 MVA) operating between 427.6 MVAR lagging and 170 MVAR leading. The cooling system consists of stator water cooling and generator hydrogen cooling equipment, each sized to support the generator at nameplate rating. The exciter has the capability to support operation within its nameplate rating and within the capability curve of the machine for the leading and lagging case of MVAR production. The main generator relaying protection scheme is not affected by the machine operation at the power uprate condition.

The staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 1124.9 MVA is below the maximum main generator design rating of 1125.6 MVA and, therefore, operating the main generator at the uprated power condition is acceptable.

<u>Main Transformer (MT)</u>

The main generator delivers its power output to two main transformers (MT 31 and 32). MT 31 is a Westinghouse transformer with a nameplate rating of 542 MVA forced-oil-air cooled (FOA) at 55 °C temperature rise. MT 32 is a General Electric transformer with a nameplate rating of 325/433/542 MVA FOA/FOA/FOA at 65 °C temperature rise. The total capacity of the MT bank is 1084 MVA. Allowing for the load of the unit auxiliary transformer, MTs will carry an apparent power of 1125.6 - 43 MVA = 1082.6 MVA.

The staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 1082.6 MVA is below the maximum main transformer design rating of 1084 MVA and, therefore, operating the main transformer at the uprated power condition is acceptable.

Isophase Bus

The isophase bus duct connects the main generator to the primary windings of the MTs and the UAT. The isophase bus system is organized into segments. The first segment runs from the generator terminals to the point where the main bus splits into the two segments that run to the two MTs. This first segment has a forced air-cooled rating of 32 kiloampheres (kA). The second segment of the main bus runs from the split to each MT. These segments have a forced air-cooled rating of 16 kA. The third segment runs from the main bus tap to the UAT. This segment has a self-cooled rating of 1.5 kA. This segment does not have a forced cooled rating. The generator operating at its nameplate rating of 1125.6 MVA at 22 kV, produces 29.54 kA. Industry standards allow generator operation at nameplate rating with a voltage range of ± 5%. At full load and 0.95 per unit voltage, the generator can produce 31.09 kA.

The staff reviewed the licensee's submittal and concluded that the currents through the segments of the isophase bus between the split and the MTs are within the bus rating for the generator at full load and 0.95 per unit voltage and, therefore, operating the isophase bus at the uprated power condition is acceptable.

Unit Auxiliary Transformer (UAT)/Station Auxiliary Transformer (SAT)

The UAT/SAT nameplate rating is 22/6.9 kV, 43MVA FOA at 55°C, three-phase, 60 Hz. The UAT supplies power to BOP systems under normal operating conditions. The BOP systems affected by the power uprate are the feedwater system, the condensate system and the heater drains system. The analysis of these systems at the 1.4% power uprate level produced new pump operating points. The feedwater pumps have non-electrical drivers so their new operating point does not effect the station electrical distribution system. There is a net load increase of 55 HP based on operation with three condensate pumps and two heater drain pumps. The increase in house load resulting from the change in operating points for the affected BOP pumps will increase house load by about 0.11%. These loads will be energized by the SAT on the occurrence of a fault.

The staff reviewed the licensee's submittal and concluded that the increase in house loads resulting from the 1.4% power uprate is below the maximum UAT/SAT design rating and, therefore, operating the UAT/SAT at the uprated power condition is acceptable.

Motor Driven Pumps

The electrical equipment that supports the mechanical systems are typically motors, cables and circuit breakers. The licensee has determined that some medium voltage motors on non-safety-related 6,900 V switchgear have revised operating points. The condensate pumps, rated at 3000 HP each, experience a brake horsepower increase from 2575 to 2600 BHP. The heater drain pumps, rated at 1000 HP each, experience a brake horsepower decrease from 940 BHP to 930 BHP. As the existing motor drives will operate at a brake horsepower less than the design rating during full load conditions at the 1.4% power uprate, no motor replacements will be required at the 1.4% power uprate.

The staff has reviewed the licensee's submittal and concluded that the brake horsepower of the condensate and heater drain pumps remain below the design rating and the design is, therefore, acceptable.

3.4.6 Summary

The NRC staff has evaluated the effect of power uprate on the necessary electrical systems and environmental qualification of electrical components. Results of these evaluations show that the increase in a core thermal power would have negligible impact on the grid stability, SBO, or the environmental qualification of electrical components. This is consistent with GDC-17 to Appendix A to 10 CFR Part 50, 10 CFR 50.63, and 10 CFR 50.49. The proposed changes are, therefore, acceptable.

3.5 System Design

3.5.1 Reactor Coolant System

The RCS operating conditions are changed slightly at uprated power. The steady-state RCS pressure (2235 psig), no-load RCS temperature (547 °F), and RCS design flows have not changed. However, the T_{cold} is decreased from 542.6 °F to 541.9 °F and T_{hot} is increased from 600.4 °F to 600.8 °F. Therefore, the RCS temperatures associated with the power uprate are still within the bounds of the original design temperature of 650 °F for RCS and 680 °F for pressurizer. Sufficient core cooling under power uprate conditions is verified by various plant transient and safety analyses. The performance of natural circulation cooldown was analyzed with 2% power measurement uncertainty with acceptable results which is applicable to the uprated power conditions. The staff finds that the changes of RCS operating parameters associated with power uprate are acceptable based on the acceptable results of the safety analyses addressed in Section 3.2.

3.5.2 Safety Injection System (SIS)

The adequacy of the SIS during the injection and sump recirculating phases following a LOCA was verified in the current licensing analysis at the current power level with the existing 2% uncertainty allowance and is applicable to the uprated power conditions. For the non-LOCA events, the performance of SIS is verified by various safety analyses performed in support of the power uprate. There are no system modifications required to support power uprate. The staff agrees with the licensee's assessment based on the acceptable results of the safety analyses addressed in Section 3.2.

3.5.3 Residual Heat Removal (RHR) System

The licensee has calculated the ability of the RHR system to achieve cold shutdown within 17 hours assuming both RHR trains operable and within 69 hours assuming only one RHR train operable under the power uprate conditions. The results of the licensee's calculation confirms that the RHR cooldown capacity meets the 72-hour Appendix R requirement, and the normal plant cooldown time changes do not affect plant safety. Based on its evaluation, the licensee has concluded that system modifications are not required to accommodate the power uprate. The staff has reviewed the licensee's submittal and agrees with the licensee's assessment.

3.5.4 SFP Storage and Cooling

The design basis for the SFP cooling system includes the capability to maintain the SFP temperature below 150 °F following the discharge of 76 fuel assemblies, and below 200 °F following the discharge of a full core. The design criteria also required that the minimum time for the SFP to boil in the event of loss of SFP cooling for the 76 fuel assemblies discharge case is 8.5 hours and 49.2 minutes for the full core offload case.

For the 76 fuel assemblies discharge case, the licensee stated that for the 1.4% power uprate condition the expected decay heat rate is 17.73 MBtu/hr which is greater than the SFP design value of 17.48 MBtu/hr. However, the maximum SFP temperature is 148.9 °F, which is below the 150 °F limit. The time for the SFP to boil in the event of a loss of SFP cooling is estimated to be 8.4 hours, which is a slight decrease from the FSAR value of 8.5 hours. Although the decay heat rate and the minimum time to boil exceeds the SFP design limits, the criterion that the maximum SFP temperature remain below the 150 °F limit is satisfied. The licensee also stated that the decrease in minimum time to boil is not significant compared to the time available to take action. In addition, the heat rates used to calculate the minimum time to boil are very conservative and the results are acceptable.

For the full core offload case, the licensee stated that for the 1.4% power uprate condition the expected decay heat rate is 30.93 MBtu/hr, which is less than the SFP design value of 35 MBtu/hr. The maximum SFP temperature is estimated to be 184.6 °F, which is below the design limit of 200 °F. The minimum time to boil is estimated to be 55.7 minutes, which is greater than the design limit of 49.2 minutes.

The licensee concluded that for the SFP cooling system there are no limitations in the existing design that would preclude the 1.4% power uprate. Component parameters are bounded by original design equipment ratings or by the original design considerations for off-normal operation, and the specific heat load, heat-up rate and heat-up time values given in FSAR Section 9.3 will be revised to reflect the design changes.

The NRC staff agrees with the licensee’s conclusions.

3.5.5 Balance of Plants (BOP) Systems and Motor-Operated Valves (MOVs)

The licensee stated that the IP3 BOP systems were reviewed for potential effects due to the 1.4% power uprate to 3067.4 MWt reactor core power. The BOP systems that could potentially be affected by the 1.4% power uprate are the:

- Main Steam and Steam Dump System (SDS)
- Condensate and Feedwater Systems (C&FS)
- Condenser/Circulating Water
- Extraction Steam System
- Feedwater Heaters and Drains
- Service Water System (SWS)
- Component Cooling Water System (CCWS)
- Containment Cooling and Filtration (CC&F) Systems
- Other Heating, Ventilation and Air Conditioning (HVAC) Systems
- Instrumentation and Controls (I&C)
- Main Turbine

The licensee evaluated the adequacy of the BOP systems based on comparing the existing design bases parameters with the uprated input parameters in Table 2-1 of the Application Report for the core power uprate conditions. The BOP piping systems evaluated for the power uprate are the main steam, condensate and feedwater, extraction steam, feedwater heaters and drains, service water, component cooling water, containment cooling and filtration, spent fuel pool cooling, and main turbine systems. As a result, the licensee concluded that the existing design basis analyses, using maximum differential temperatures and pressures for normal operation and worst-case conditions, for the BOP piping, pipe supports, and components remain bounding for the uprated power level of 3067.4 MWt at IP3.

The IP3 BOP systems were evaluated based on actual heat balance data at current power conditions, and updated to reflect operating conditions expected for the 1.4% power uprate. The system components (turbine, pumps, valves, feedwater heaters, fans, etc.,) were evaluated for conformance with the original design requirements considering thermal-hydraulic performance, decay heat loads, equipment design capacity, piping integrity, and I&C system design requirements.

The licensee concluded that for the 1.4% power uprate the new operating parameters for the existing BOP system components were either bounded by original design equipment ratings or by the original design considerations for off-normal operation including DBAs. For those BOP systems and components required to operate during and following design bases accidents, the required post-accident functions were analyzed by the licensee for at least 102% of rated thermal power. These analysis bound the 1.4% power uprate conditions.

The NRC staff concludes that the BOP system evaluations performed by the licensee identified no limitations in the existing system or component design that would preclude the 1.4% power uprate.

The licensee also reviewed the programs, components, structures, and generic letter issues as they pertain to the power uprate. In the Application Report, the licensee confirmed that there are no changes to the IP3 MOV program as a result of the 1.4% power uprate. Safety related valves were not found to be impacted by the 1.4% power uprate and are, therefore, acceptable. This determination was confirmed by verifying that changes in system operating temperature, pressure and flow rate were bounded by the design basis requirements. Additionally, in the Application Report, the licensee assessed the impacts of the 1.4% power uprate on the programs covering NRC Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” dated June 28, 1989, and GL 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves,” dated September 18, 1996, and found them to be acceptable.

The licensee reviewed its current evaluation of GL 95-07,”Pressure Locking and Thermal Binding [PLTB] of Safety- Related Power-Operated Gate Valves,” dated August 17, 1995, for valves that were listed in the GL 95-07 as being modified to eliminate the potential for PLTB and found them to remain bounding for the 1.4% power uprate. In the letter dated September 13, 2002, responding to the staff’s request for additional information, the licensee indicated that the existing evaluation for GL 96-06, “Assurance of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions,” dated September 30, 1996, was performed at 102% of the current power and is, therefore, bounding for the proposed power uprate of 101.4% rated power level. On the basis of the above review, the staff concurs with the licensee's conclusions that the power uprate will have no adverse effects on the performance of safety-related valves and that conclusions reached based on implementation of provisions in GL 95-07, GL 96-06, GL 89-10 and GL 96-05 programs remain valid.

As a result of the above evaluation, the staff concludes that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and continue to satisfy the design-basis requirements for the proposed 1.4% power uprate.

3.5.6 Radioactive Waste Processing Systems

Since reactor operation at the increased power level after the 1.4% power uprate will increase the release rate of radioactive isotopes into the reactor coolant, the activity levels associated with liquid and gaseous effluents will increase proportionally. However, the volume of solid waste would not be expected to increase proportionally since equipment performance and system operation is not appreciably changing. The radioactive wastes are processed through the solid, liquid, and/or gaseous radioactive waste system. The licensee states that the wastes generated will be processed within the plant and there would be minimal effects from the additional waste generation.

Based on this statement and experience gained from the review of power uprate applications for similar PWR plants, the staff concludes that the solid, liquid, and gaseous radioactive waste systems are acceptable for the plant power uprate operations.

3.5.7 Containment System

The licensee reviewed the existing containment integrity analysis to ensure the maximum pressure inside the containment would not exceed the containment design pressure if a design basis LOCA or MSLB inside containment should occur during plant operation. The review also established the pressure and temperature for environmental qualification and operation of safety-related equipment located inside the containment. The results of the review follow.

Containment Integrity Analysis - Loss-of-Coolant Accident

The licensee states that the current mass and energy release data for input into the containment response analysis assumed a core thermal power of 3025 MWt, plus an additional 2% power measurement uncertainty allowance. The results of this analysis bounds the power uprate level of 3067.4 MWt , a 1.4% uprate with a 0.6% uncertainty. Therefore, the mass and energy release data for the LOCA bound the power uprate conditions, and the peak LOCA containment pressure and temperature will not be affected by the power uprate.

The licensee also conducted a short-term LOCA mass and energy release calculation to support the reactor cavity and loop subcompartment pressurization analyses. These analyses are performed to ensure the structural integrity of walls in the immediate proximity of the postulated break location to withstand the short pressure pulse within the region resulting from the LOCA. The licensee found that the critical flow correlation used in the mass and energy releases for this analysis provides an increase in the mass and energy release for a slightly lower fluid temperature. At the 1.4% power uprate level, the minimum full power RCS cold leg temperature can be 3.4 °F lower than that used in the design basis evaluation. The licensee states that the decrease in minimum RCS cold leg temperature is offset by the reduction in mass and energy releases from the smaller primary system nozzle breaks than assumed in the original design basis. Therefore, the current short-term subcompartment pressurization analysis is unaffected by the power uprate.

Containment Integrity Analysis - Main Steam Line Break (MSLB)

The licensee states that mass and energy releases and containment integrity were evaluated for an MSLB based on the 1.4% power uprate conditions. As indicated by the values for design parameters for the 1.4% power uprate in Table 2-1 of the Analysis Report, the licensee determined that the parameters either remain unchanged or are bounded by the current analysis values in the mass and energy calculations. Therefore, the results of the containment integrity analysis for the MSLB is not affected by the 1.4% power uprate.

Based on review and assessment of the information provided in the licensee's submittal, the staff finds acceptable the licensee's conclusion that the peak LOCA, as well as MSLB containment pressure and temperature, will not be affected by the power uprate, and that the containment integrity analysis at the proposed uprated power is bounded by current LOCA and MSLB analysis.

3.6 Other Areas of Review

3.6.1 10 CFR Part 50, Appendix R, Fire Protection

The licensee stated that the Emergency Lighting and RCP Oil Collection Systems are not affected by the 1.4% power uprate. However, the 1.4% power uprate will affect plant cooldown times. The licensee performed a single-train cooldown analysis to support the worst-case scenario for the 10 CFR Part 50, Appendix R fire hazards analysis. The worst-case scenario included in part: reduced RHR system cooldown capacity due to RHR pump miniflow, and increase heat load on the component cooling water system due to additional heat generated in the spent fuel pool due to the 1.4% increased in core power. The analysis indicated that the RHRS was capable of achieving RCS cold shutdown (below 200 °F) in less than 72 hours after reactor shutdown (see Section 6.1.3 of the Application Report).

For a postulated fire with a LOOP, the EDGs are the preferred source of AC power for the safe shutdown systems. An Appendix R diesel generator is also available at IP3 to enhance the plant's alternate shutdown capability. The licensee stated that the Appendix R diesel generator loads are not affected by the 1.4% power uprate.

Since the 72-hour cooldown requirement is maintained for the power uprate conditions and the Appendix R diesel generator loads are not affected by the 1.4% power uprate, the NRC staff concludes that the safe shutdown capability, with regard to Appendix R requirements, is not affected by the power uprate.

3.6.2 Human Factors

The NRC staff reviewed the following operator performance topics discussed in the licensee's application.

3.6.2.1 Plant Procedures

The licensee stated that the power uprate has no significant effect on plant operating procedures. Where changes are required, the procedures will be revised or updated in a manner consistent with any other plant modification. The licensee also stated that procedure limitations on power operations due to BOP equipment unavailability, such as updated neutron flux trip setpoints with inoperable MSSVs, will be revised as necessary to account for the increase in core power to 3067.4 MWt. Those procedures required for the operation and maintenance of Caldon LEFM Check System are being revised as necessary to reflect installation of the Check System. Specific actions to be taken when the Caldon LEFM Check System is inoperable is addressed in section 3.1 above and will be included in the IP3 Technical Requirements Manual.

The NRC staff finds that the licensee's response is satisfactory because the procedures will be revised to incorporate the Caldon LEFM Check System prior to implementation of the power uprate. In addition, the licensee will treat plant procedure changes due to power uprate in a manner consistent with any other plant procedure change.

3.6.2.2 Changes to Risk-Important Operator Actions Sensitive to Power Uprate

The licensee stated that ESF system design and setpoints and procedural requirements already bound the proposed power uprating. The responses of the reactor operators to any event will be essentially unaffected by a change in rated thermal power (RTP).

There will be minimal impact on alarms, and procedural requirements for 1.4% uprating. The Caldon LEFM Check System will have alarms in the control room to alert operators of conditions that impair its availability and accuracy. No other alarm impacts are expected. It is not anticipated that any existing alarms will be modified or deleted. Alarms will be re-calibrated as necessary to reflect small setpoint changes. However, no significant or fundamental setpoint changes are anticipated. Also, the operator response to existing alarms is anticipated to remain as before.

When the power uprate is implemented, the nuclear instrumentation system will be adjusted to indicate the new 100% RTP in accordance with TS requirements and plant administrative controls. Since the power uprate is predicated on the availability of the system, procedural guidance will be implemented to facilitate operation when the Caldon LEFM Check System is unavailable. The reactor operators will be trained on the changes in a manner consistent with any other design modification.

The power uprate will be reflected in the plant simulator. These changes should be virtually transparent to the reactor operators.

The NRC staff finds that the licensee's response is satisfactory because the licensee has adequately addressed the question of operator actions sensitive to the power uprate and shown that the responses of the reactor operators to any event will be essentially unaffected by 1.4% increase in the rated thermal power. The licensee will implement procedures and guidance as required for operator actions when Caldon LEFM Check System is unavailable.

3.6.2.3 Changes to the Safety Parameter Display System (SPDS)

The licensee stated that only process parameter scaling changes will be made, as required, to the Qualified Display Parameter System (QDPS), and that there are no other impacts to the QDPS due to 1.4% uprate.

The NRC staff finds that the licensee's response is satisfactory because the licensee will identify and make the necessary scaling changes to the QDPS as a result of the power uprate.

3.6.2.4 Changes to the Operator Training Program and the Control Room Simulator

The licensee's response to this question is included in the discussion in Section 3.6.2.2. The NRC staff finds the licensee's response satisfactory, because the licensee has adequately described how the changes to operator actions will be addressed by training and how the simulator will accommodate the changes.

3.6.2.5 Human Factors Evaluation Summary

The NRC staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The NRC staff further concludes that the power uprate will not adversely affect simulation facility fidelity, operator performance, or operator reliability.

3.7 Technical Specification Changes

The licensee submitted proposed TS changes to support safe operations at an uprated power level of 3068 MWt. The following is the staff review of the TS changes.

TS Section 1.1

The definition of RTP is revised to reflect the increase from 3025 MWt to 3067.4 MWt. Because this TS change reflects the actual proposed change in the plant and is consistent with the results of the licensee's supporting safety analyses, the NRC staff finds the change acceptable.

Safety Limits -TS 2.1.1, Figure 2.1-1

IP3 TS Figure 2.1-1 shows the loci of points of RPV inlet temperature versus rated thermal power at various RCS pressures for which the calculated DNBR is no less than the safety limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The licensee proposed change to Figure 2.1-1 reflects the following: (1) the effect that the 1.4% power uprate has on the exit boiling limit and more conservative axial offset control, and (2) the value for 100% rated thermal power is changed from 3025 to 3067.4 MWt.

The licensee has demonstrated in its supporting safety analyses that, with the plant protection system functioning, the results of transient analyses show that the proposed safety limits are not violated. Therefore, the NRC staff finds the proposed change acceptable.

High Steam Flow ESF Actuation Allowable Values, TS Table 3.3.2-1

The licensee proposed changing the allowable value for Functions 1.f, “Safety Injection High Steam Flow in Two Steam Lines Coincident with T_{avg} - Low,” 1.g, “Safety Injection High Steam Flow in Two Steam Lines Coincident with Steam Line Pressure - Low,” 4.d, “Steam Line Isolation High Steam Flow in Two Steam Lines Coincident with T_{avg} - Low,” and 4.e, “Steam Line Isolation High Steam Flow in Two Steam Lines Coincident with Steam Line Pressure - Low,” in TS Table 3.3.2-1, “Engineered Safety Feature Actuation System Instrumentation.” The allowable value for these functions is a variable that increases linearly and is dependent on the turbine first stage pressure that corresponds to the percentage of steam flow load. As stated in Note (c) to Table 3.3.2.1, the current allowable value is "Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 110% full steam flow at 100% load, and corresponding to 110% full steam flow above 100% load. Time delay for SI $\leq$ 6 seconds." The proposed revision would replace the maximum value of "110% of full steam flow at 100% load" with "120% of full steam flow at 100% load."

On page 14.2-29, the IP3 FSAR states that in the IP3 TS the high steam flow safety injection setpoint is a function defined as follows: a ΔP corresponding to 54% of full steam flow between 0% and 20% load and increasing linearly to a ΔP corresponding to 110% of full steam flow at full load. A 24% uncertainty in steam flow was added to account for channel errors and adverse environmental errors. These allowable values for the high steam flow setpoints are reflected in the current TS Table 3.3.2-1 related to safety injection and main steam isolation actuation. The licensee proposed a change to increase the upper bound setpoint from 110% to 120% by reducing the margin for uncertainties from 24% to 14%. The reason for the proposed change is to reduce potential inadvertent actuation of safety injection system and main steam isolation. The NRC staff finds that the proposed changes to this instrumentation setpoint allowable value reflect the 1.4% power uprate and that it was developed in accordance with the IP3 instrument setpoint methodology. The staff also finds that the proposed change provides sufficient margin for uncertainties remaining in the setpoints and, therefore, is acceptable.

TS Figures: 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.12-1, 3.4-12-2, 3.4-12-3, and 3.4.12-3

For the various limitation curves for heatup, cooldown, leak testing, and low temperature overpressurization protection (LTOP) listed above, the licensee proposed to change the applicable service period from 16.2 to 16.17 Effective Full Power Years (EFPY) to reflect changes due to the 1.4% power uprate.

The P-T limits reflected in the proposed TS Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 and the LTOP limits reflected in TS Figures 3.4.12-1, 3.4-12-2, 3.4.12-3, and 3.4.12-4 correctly represent the revised range of validity of the corresponding curves and therefore, are acceptable.

Operable Main Steam Safety Valves Versus Applicable Neutron Flux Trip Setpoints, TS Table 3.7.1-1

The licensee’s proposed change will decrease the allowable values for the neutron flux trip setpoint during operation with one or more MSSVs inoperable. The licensee indicated that these changes are appropriate due to the slight increase of rated thermal power level due to the proposed 1.4% power uprate. The changes to Figure 3.7.1-1 will provide more conservative trip setpoints at different MSSV inoperable conditions. The staff finds the licensee's proposal acceptable because the changes are in the conservative direction from the current TS at IP3.

TS Bases Associated with the Proposed TS Changes

The licensee proposed changes to the following TS Bases:

1. B 3.3.1, "RPS Instrumentation," changes the value of uncertainties from 2% to 0.6%
2. B 3.3.2, ESFAS Instrumentation," changes the allowable high steam flow from 110% to 120% of full steam flow at full load
3. B 3.6.6, "Containment Spray System and Containment Fan Cooler System," B 3.7.1, "MSSVs," and B 3.7.6, "CST," changes the analyses power level from 102% to 100.6%
4. B 3.7.1 Changes NSSS power rating (including pump heat) from 3037 to 3081.4 MWt

These proposed changes are consistent with the TS changes discussed above. Therefore, the staff finds the proposed changes to the TS Bases acceptable.

In summary, the NRC staff finds that the proposed TS changes as discussed in Section 3.0 of this evaluation adequately reflect the results of the acceptable supporting analysis and, therefore, concludes that the proposed TSs are acceptable for the power uprate applications.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 45565). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. ENO Letter, Robert J. Barrett to NRC, “Request for Licensing Amendment for 1.4% Measurement Uncertainity Recapture Power Uprate,” dated May 30, 2002.

2. ENO Letter, Robert J. Barrett to NRC, “Reply to Request for Additional Information Regarding Proposed Licensing Amendment for 1.4% Measurement Uncertainity Recapture Power Uprate,” dated September 13, 2002.

3. ENO Letter, Robert J. Barrett to NRC, “Reply to Request for Additional Information Regarding Proposed Licensing Amendment for 1.4% Measurement Uncertainity Recapture Power Uprate,” dated November 6, 2002.

4. ENO Letter, Robert J. Barrett to NRC, “Reply to Request for Additional Information Regarding Proposed Licensing Amendment for 1.4% Measurement Uncertainity Recapture Power Uprate,” dated November 20, 2002.

5. Westinghouse Report WCAP–10263, “A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant,” dated January 1993.

6. Caldon, Inc. Topical Report ER-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System,” NRC approved March 8, 1999.

7. Caldon, Inc. ER-157P Topical Report, Supplement to Topical Report ER-80P, “Basis for a Power Uprate with the LEFM Check or LEFM CheckPlus System,” NRC approved December 20, 2001.

8. Caldon, Inc. ER-160P Topical Report, Supplement to Topical Report ER-80P, “ Basis for a Power Uprate with the LEFM Check System,” NRC approved January 19, 2001 as part of the Watts Bar SER approval.

Principal Contributors: I. Ahmed
M. Hart
C. Khan
C-Y Liang
L. Lois
B. Marcus
T. McLellan
P. Milano
N.Trehan
C-I Wu

Date: November 26, 2002

cc:

Mr. Jerry Yelverton
Chief Executive Officer
Entergy Operations
1340 Echelon Parkway
Jackson, MS 39213

Mr. Robert J. Barrett
Vice President - Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
P. O. Box 308
Buchanan, NY 10511-0308

Mr. Dan Pace
Vice President Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. James Knubel
Vice President Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Joseph DeRoy
General Manager Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
P. O. Box 308
Buchanan, NY 10511-0308

Mr. John Kelly
Director - Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Ms. Charlene Faison
Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Harry P. Salmon, Jr.
Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. James Comiotes
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
P.O. Box 308
Buchanan, NY 10511-0308

Mr. John McCann
Manager, Licensing and Regulatory Affairs
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 2
295 Broadway, Suite 1
P. O. Box 249
Buchanan, NY 10511-0249

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
295 Broadway, Suite 3
P.O. Box 337
Buchanan, NY 10511-0337

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. John M. Fulton
Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Ms. Stacey Lousteau
Treasury Department
Entergy Services, Inc.
639 Loyola Avenue
Mail Stop: L-ENT-15E
New Orleans, LA 70113

cc:

Mr. William M. Flynn, President
New York State Energy, Research, and
Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. J. Spath, Program Director
New York State Energy, Research, and
Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
Electric Division
New York State Department
of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Ray Albanese
Executive Chair
Four County Nuclear Safety Committee
Westchester County Fire Training Center
4 Dana Road
Valhalla, NY 10592

Mr. Ronald Schwartz
SRC Consultant
64 Walnut Drive
Spring Lake Heights, NJ 07762

Mr. Ronald J. Toole
SRC Consultant
Toole Insight
605 West Horner Street
Ebensburg, PA 15931

Mr. Charles W. Hehl
SRC Consultant
Charles Hehl, Inc.
1486 Matthew Lane
Pottstown, PA 19465

Mr. Alex Matthiessen
Executive Director
Riverkeeper, Inc.
25 Wing & Wing
Garrison, NY 10524

Mr. Paul Leventhal
The Nuclear Control Institute
1000 Connecticut Avenue NW
Suite 410
Washington, DC, 20036

Mr. Karl Copeland
Pace Environmental Litigation Clinic
78 No. Broadway
White Plains, NY 10603

Jim Riccio
Greenpeace
702 H Street, NW
Suite 300
Washington, DC 20001